Neutronics tools and design analysis for the stellarator power reactor HELIAS

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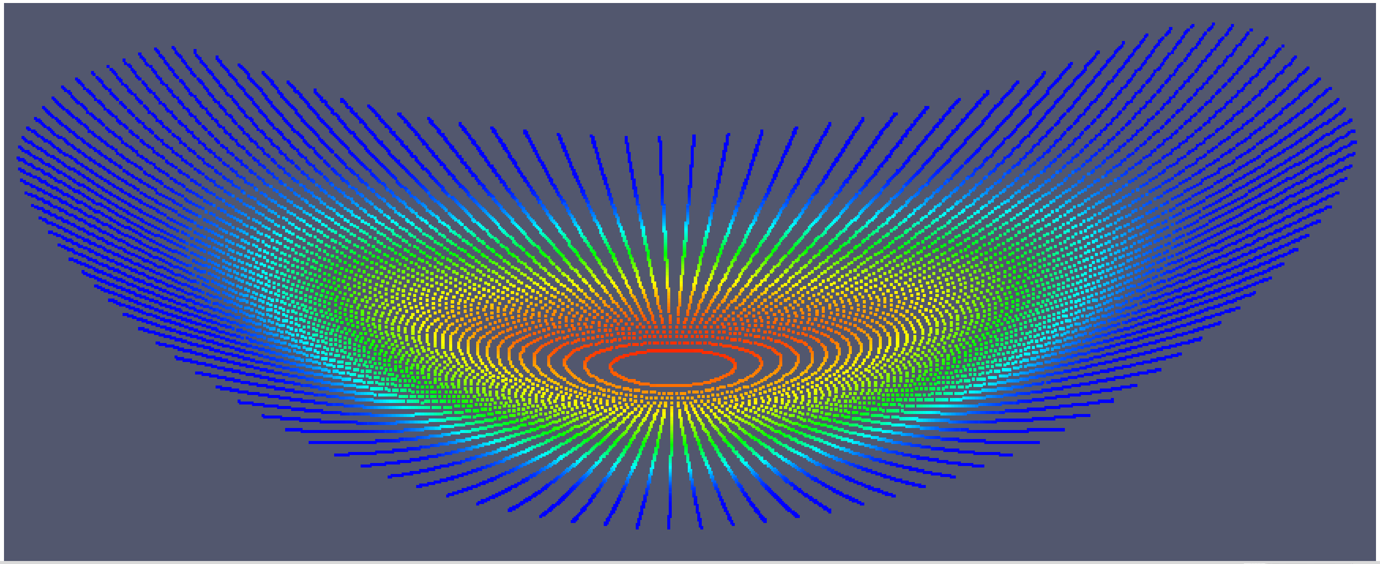

Outline

- Motivation
- Neutron source development
- CAD based geometry modeling
- Conclusion and outlook

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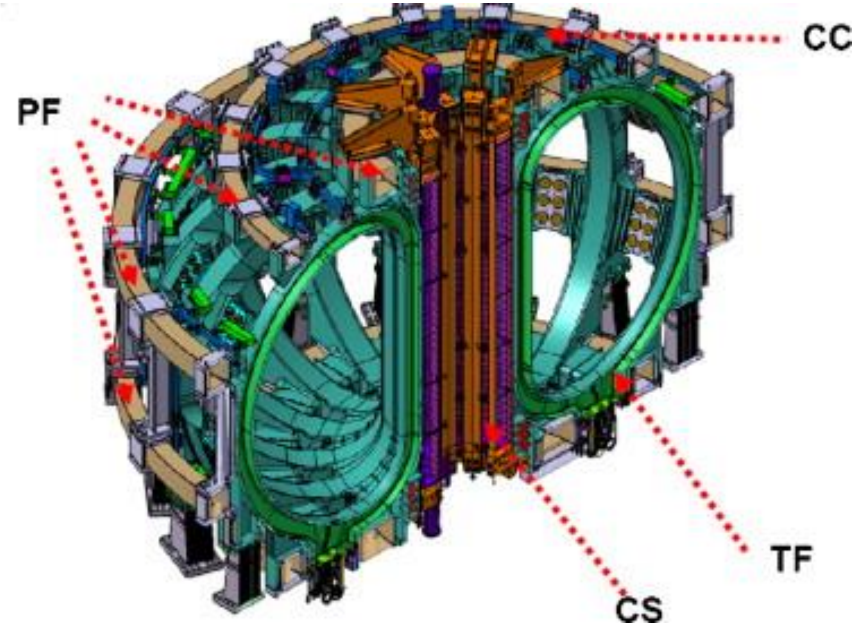
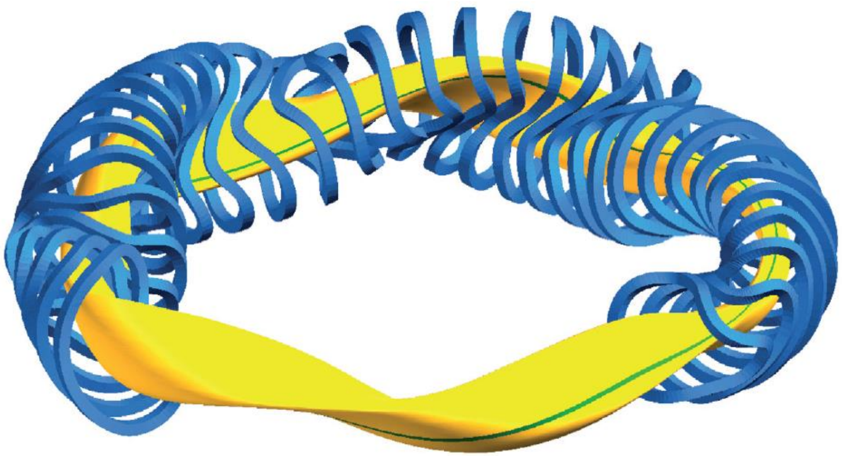
MOTIVATION

PhD research plan

- Objective: developing of a suitable computational approach for neutronic analyses of a stellarator type fusion reactor and application to design analyses of the HELIAS power reactor
- Separated into three parts:
  1. Development of a neutron source model
  2. Approaches for stellarator modelling
  3. Design analyses for the HELIAS reactor

Comparison: Tokamak and Stellarator





Tokamak

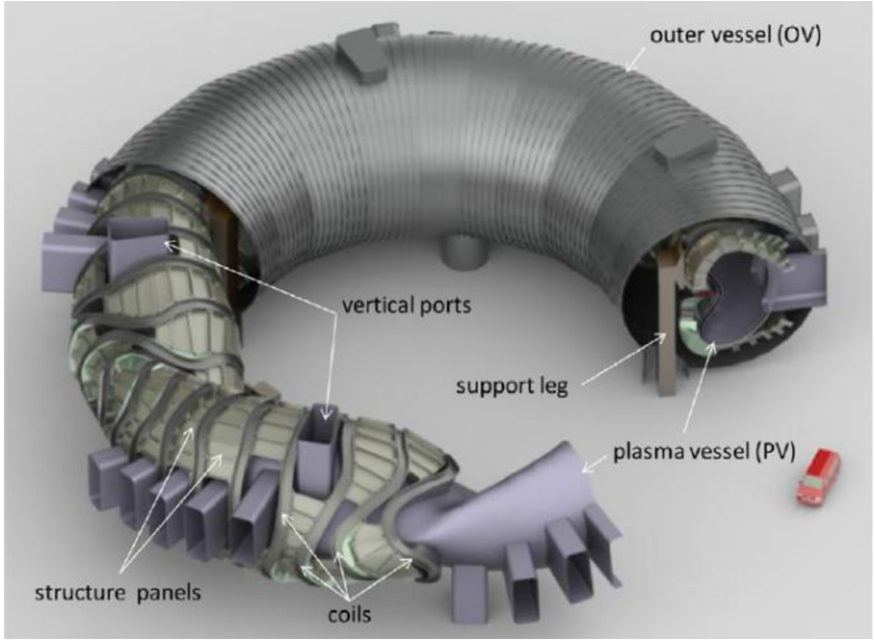
Source: [Mit09]

Stellarator

Source: [IPP13]

HELIAS in detail

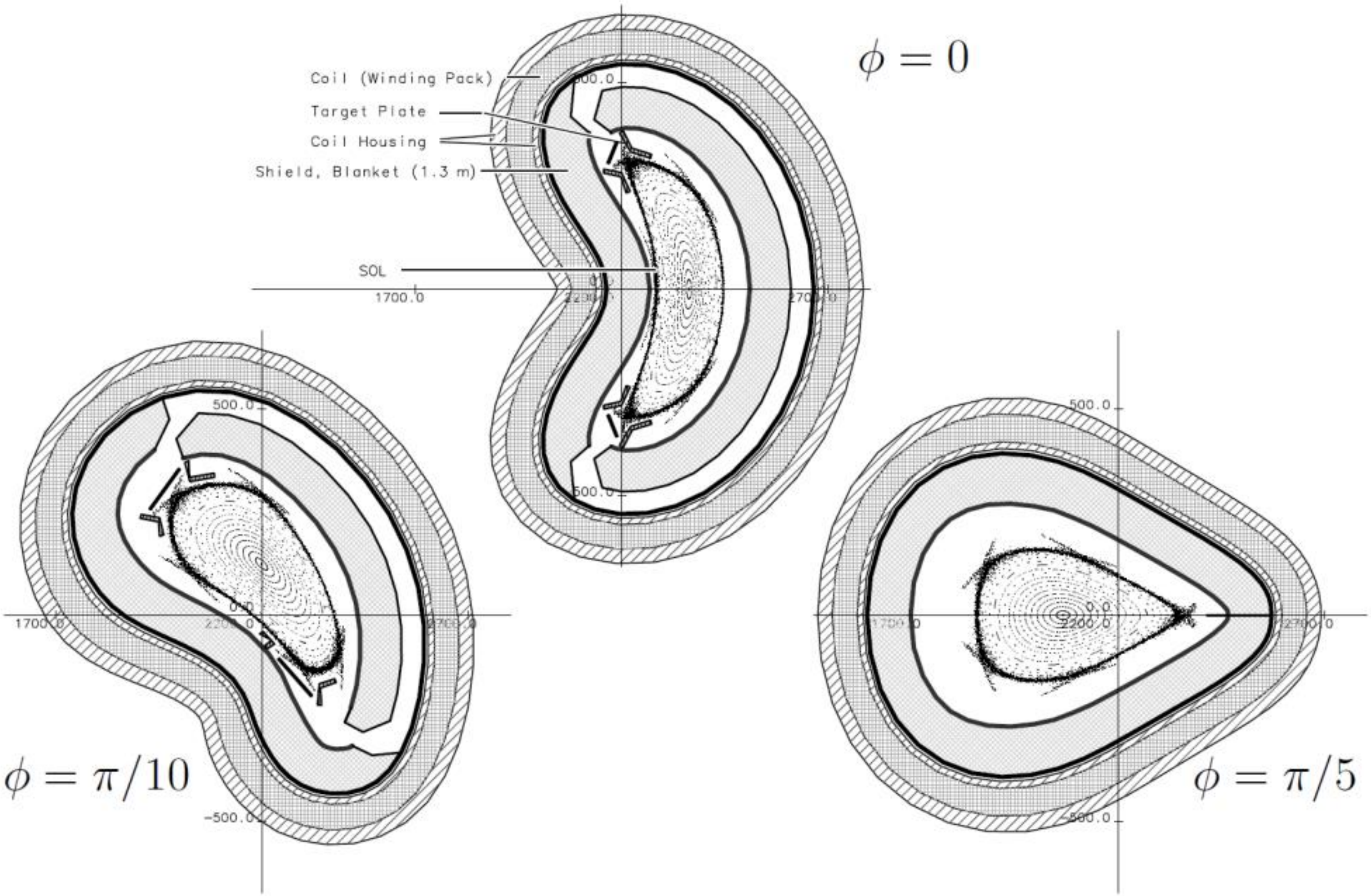
- HELIAS = **HELI**cal **A**dvanced **S**tellarator
- Upgraded version of Wendelstein 7-X
- straightforward extrapolation
- Demonstration power reactor study with D-T Fusion
- Plasma volume: ~1400 m^3
- Fusion power: ~3000 MW
- Technology for this reactor will be investigated



HELIAS 5-B
Source: [Sch13]

HELIAS cross section

- HELIAS toroidal cross sections, Poincaré Plot (Source: [Bei11])



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How to handle such a geometry in a neutron transport simulation?

- Neutrons from the plasma are interacting with the surrounding components
- Simulation should represent the geometry as good as possible
- Optimized components arrangement around the plasma chamber is needed
- Complicate to handle such a complex geometry with deterministic methods
- Statistical methods are needed to solve the problem for a complex geometry and track the particles on a microscopic level → Monte Carlo Method

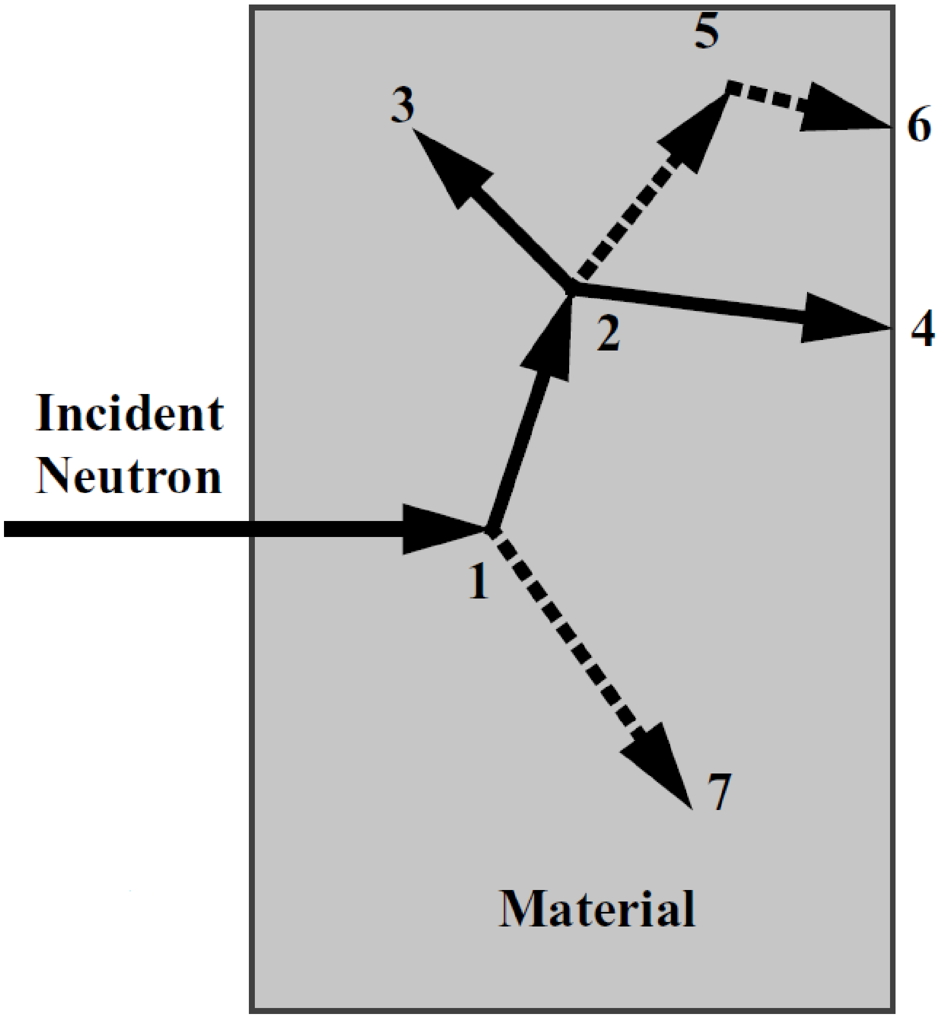
Monte Carlo Method

- Simulation of the true physical process on a microscopic level
- Probabilistic method → statistical registration of stochastically processes
- Run many histories to get many events and count them → results are statistically reliable
- Every history contributes the same weight for the end result
- Monte-Carlo (MC) Method is the superior method for fusion neutronics

Monte Carlo particle transport

Event Log

1. Neutron scatter, photon production
2. Neutron, photon production
3. Neutron capture
4. Neutron leakage
5. Photon scatter
6. Photon leakage
7. Photon capture



Source: [Tea08]

MCNP – particle transport code

- MCNP (**M**onte **C**arlo **N**-**P**article) is a general-purpose particle transport code developed by *Los Alamos National Laboratory, USA*
- Code can perform neutron, photon, electron, or coupled n/p/e transport
- Treats an arbitrary three-dimensional configuration of materials in geometric cells
- Pointwise energy dependent cross-section nuclear data files used for the simulation, e.g. Fusion Evaluated Nuclear Data Library (FENDL)

MCNP – particle transport code

Karlsruhe Institute of Technology

- User creates Input file that contains information about the problem in areas such as:
  - Geometry specification
  - Description of materials and selection of cross-section evaluations
  - Location and characteristics of the particle source
  - Type of response or quantities desired (e.g. tallies)
  - Any variance reduction techniques used to improve efficiency

Why use MCNP for fusion calculations?

- Development of the code started in the 1960th → started for nuclear fission and nowadays used for a wide range of applications
- Code is well validated in fission, fusion and accelerator field → experimental data was compared with simulation results from MCNP
- Can handle complex geometry
- Code is able to run in parallel mode
- Standard code for fusion neutronics calculations of ITER

NEUTRON SOURCE DEVELOPMENT

Basics

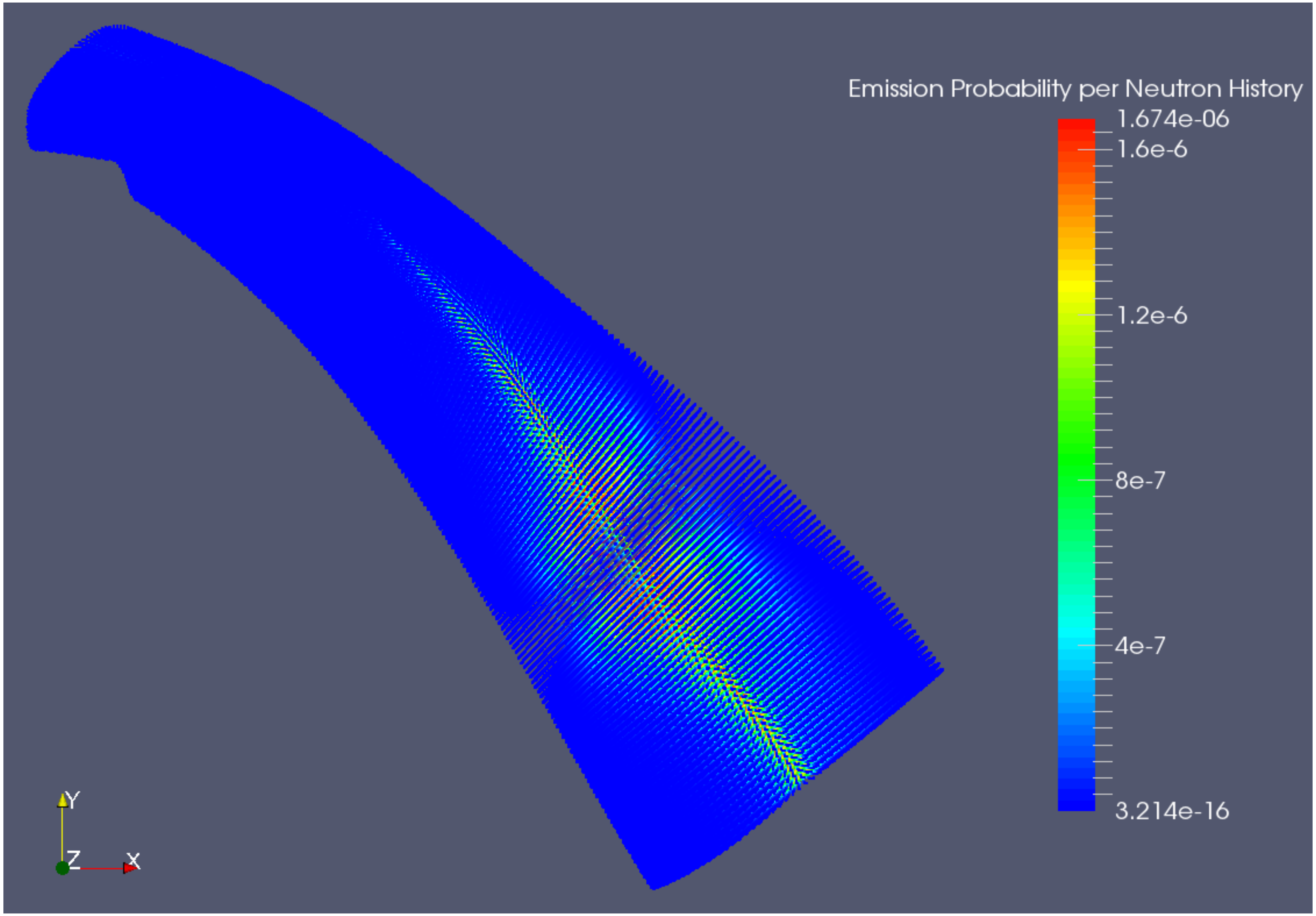
- Reaction in the plasma chamber: D + T → α + n
- Neutrons with an energy of ~14.1 MeV
- Neutronics calculations using MCNP
- Primary distribution of the (D,T) source neutrons (spatial and intensity distribution) is provided by plasma physics simulations performed by IPP Greifswald

Neutron source distribution

- Intensity per neutron history is distributed over ten orders of magnitude (10^{-6} to 10^{-16})
- Tabulated data provided by F. Warmer, IPP Greifswald

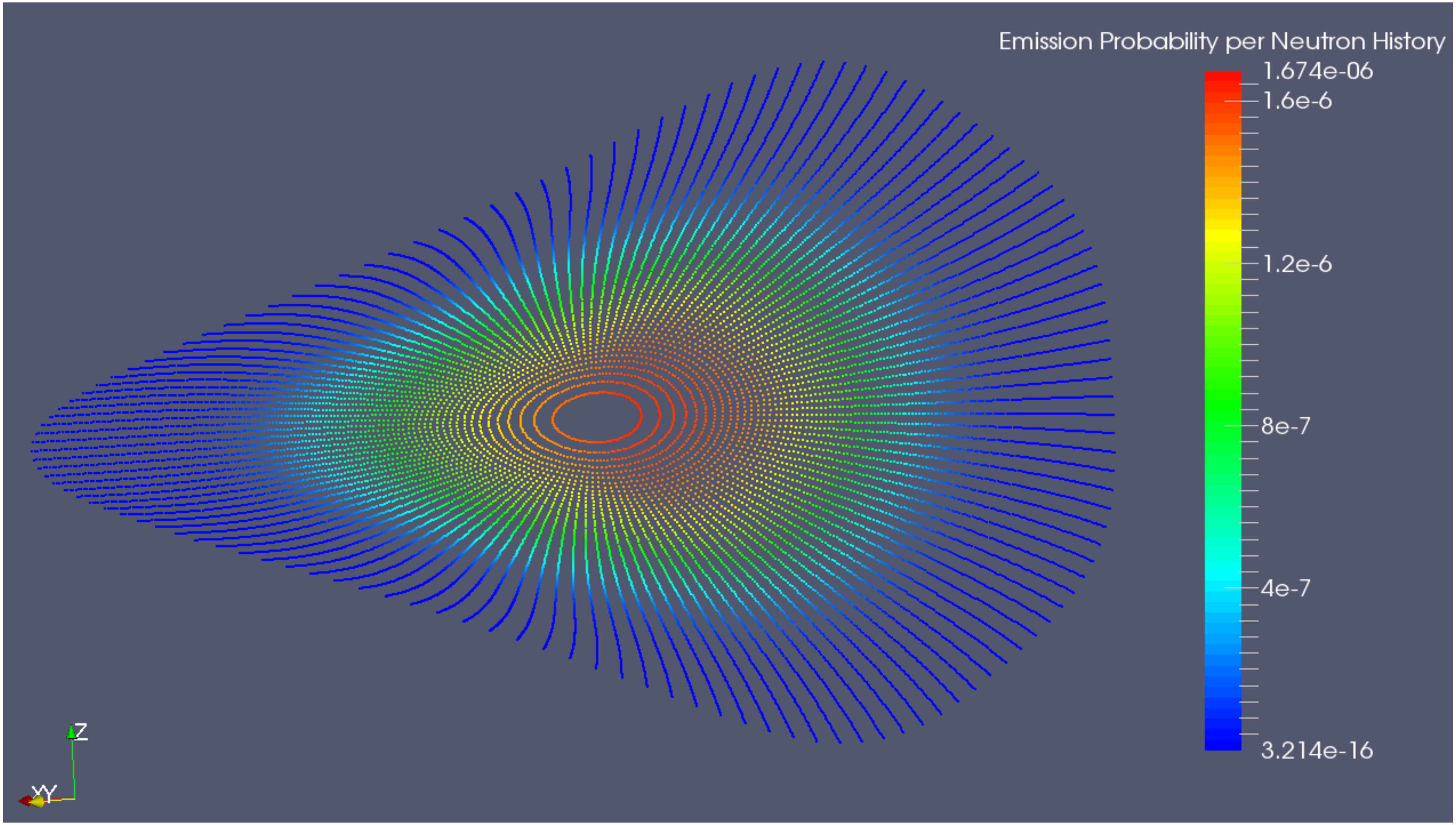
- The subsequent slides show some plots of the neutron source distribution, XYZ points, linked with their emission probability
- Data points have an approximately distance of 7cm along the main axis
- Data point density varies in the directions perpendicular to the main axis

Neutron source distribution



Emission probability of the source at the main axis of the half-field period

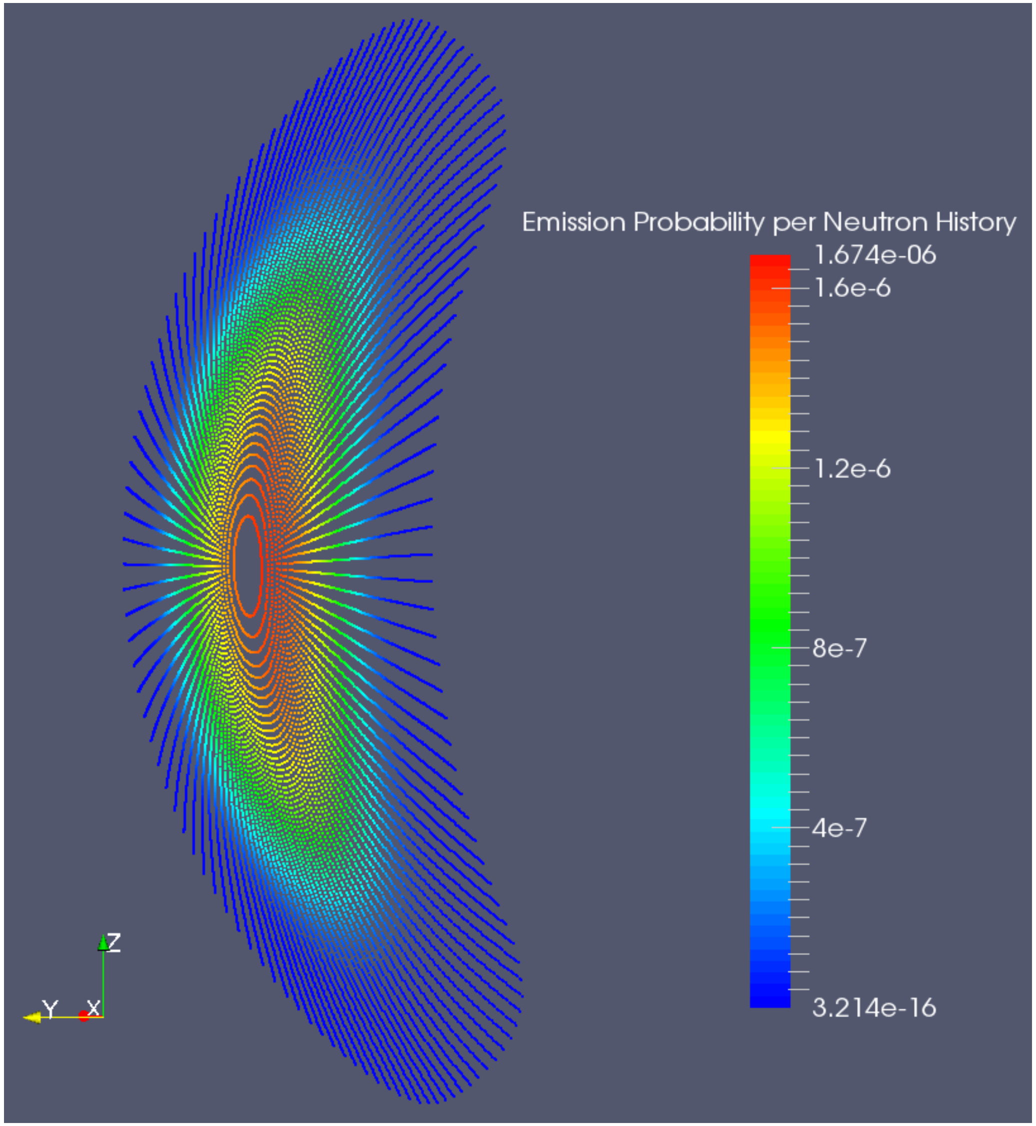
Neutron source distribution



Emission probability of the source perpendicular to the main axis

Neutron source distribution

Emission probability of the source perpendicular to the main axis



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MCNP requirements for a source description

- MCNP needs a source to perform the calculation:
  1. standard source definition of MCNP (“SDEF source”) → written in input file
  2. source subroutine → written in Fortran90 and recompilation of the code
- Requirements for the source:
  - XYZ position of the starting point
  - Particle type
  - Energy
  - Weighting
- Additional information like angular distributions or emission probabilities can also be provided in the source description

Neutron source development

- Approach:

➢ Develop dedicated source routine for neutronics calculation → plasma distribution is too complex for the standard source

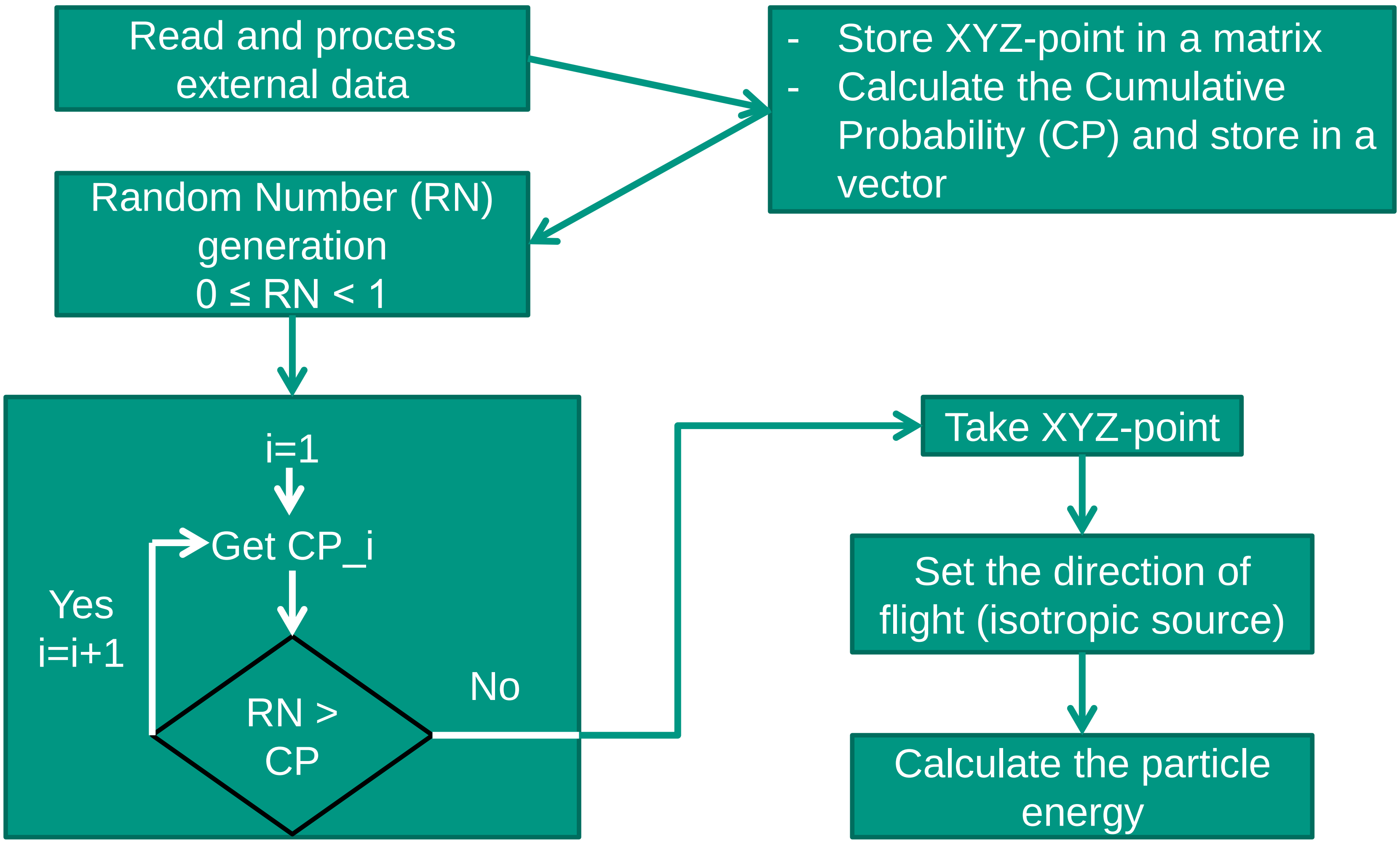
➢ Source position (X,Y,Z) and intensity stored in an external file

➢ Calculate and store cumulative probability of the intensity

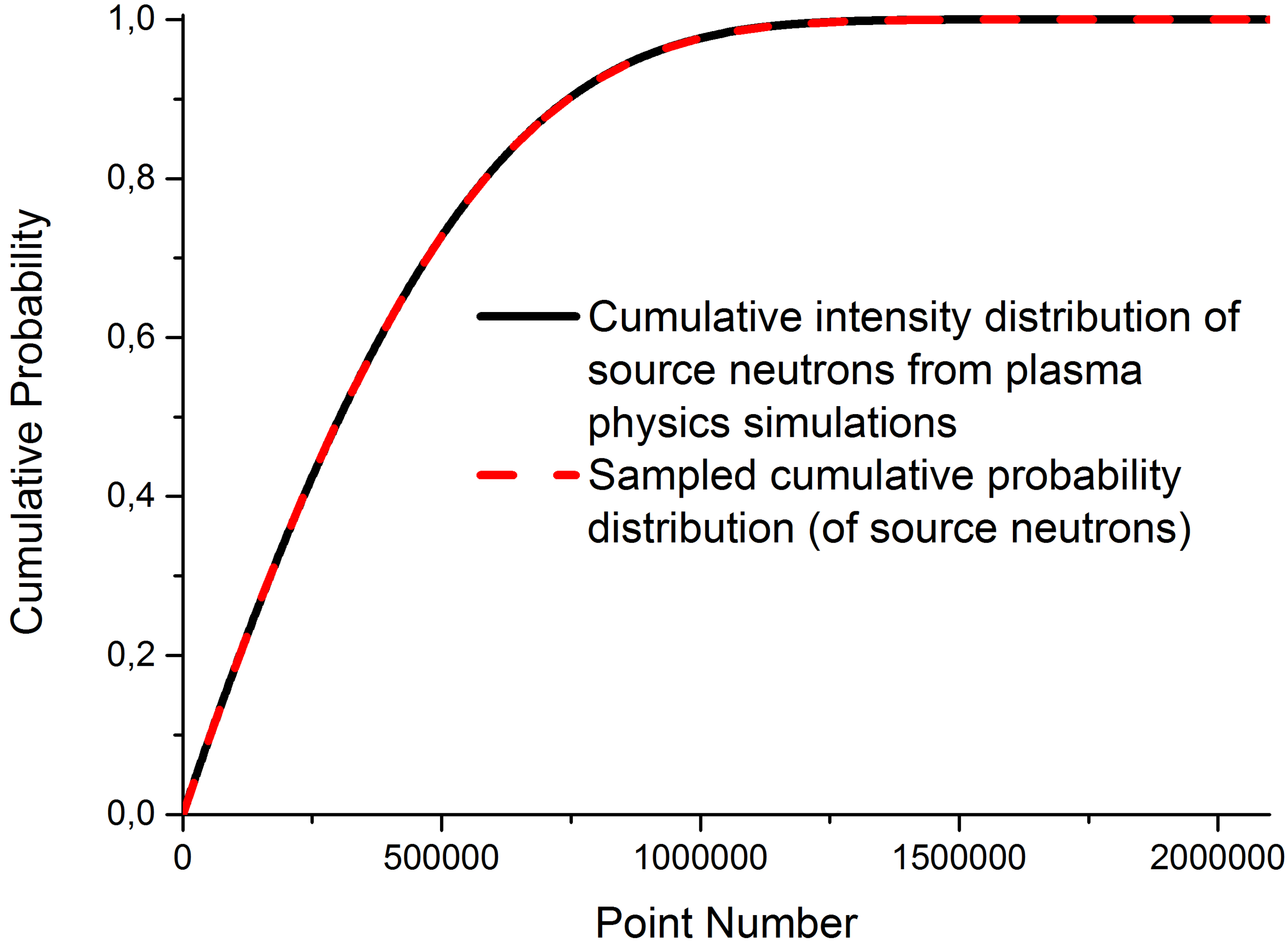
➢ Sample source position through the cumulative probabilities → regions with higher intensity have a higher sample frequency → same weight of all emitted neutrons

➢ Sample energy and direction of flight (angle)

Source point sampling



Cumulative probability of the intensity



Search algorithm

Sequential Search

- Simplest search algorithm
- Straightforward
- Expected number of comparisons is: $\frac{n+1}{2}$ (equidistant grid)
- Cost: $O(n)$

Binary Search

- Half-interval search (begin in the middle of the interval to compare with searched value)
- Needs sorted array → is done by cumulative probability of the intensities
- Cost: $O(\log(n))$

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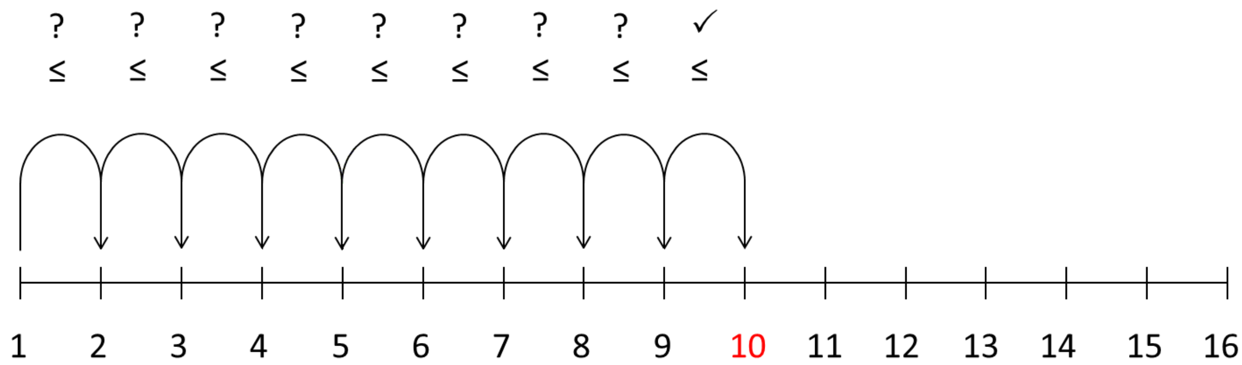
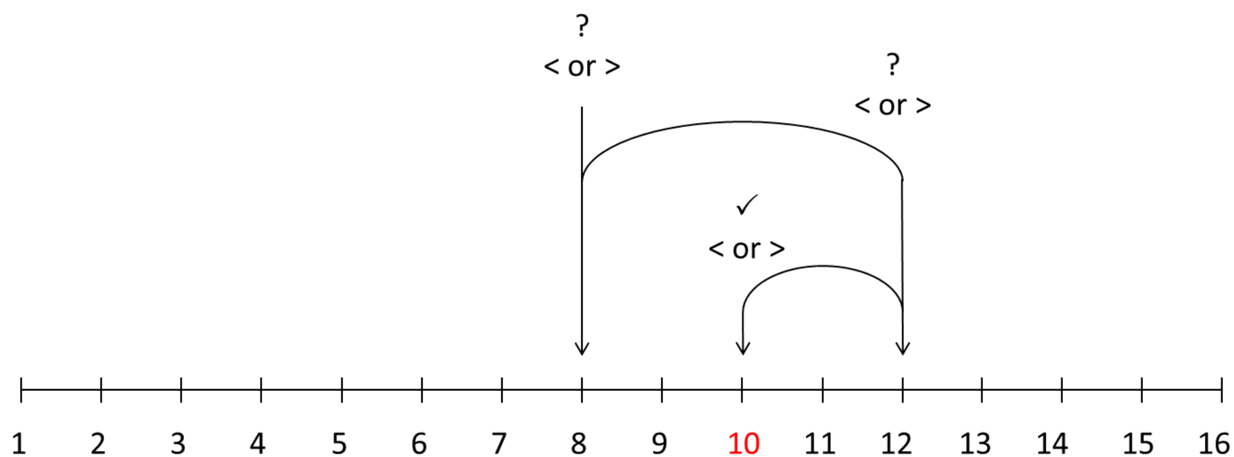
Search algorithm

Sequential Search



Binary Search



$= \frac{n}{2}$ $\quad = \frac{\left(\frac{n}{4}\right)}{2}$ $\quad = \frac{\left(\frac{n}{2}\right)}{2}$

Search algorithm comparison

- Random number generation, find this number (lower/equal) in the cumulative probability list
- Output: line number
- Same random number generator used for all the tests
- Output from both search algorithm compared → perfect accordance
- In my case, binary search ~22 times faster then sequential search (search of ~$2{,}1\times10^{6}$ elements)

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Check of the source points

- Comparison of the XYZ-points from the plasma physics and the neutron physics calculation
- In MCNP: store all sampled source points in an external file
- Used 10^9 particle, size of external file ~115 GB
- Postprocessing: generate frequency of the sampled source points → list with XYZ-points and frequency
- Compare the generated frequency with the intensity from original data file

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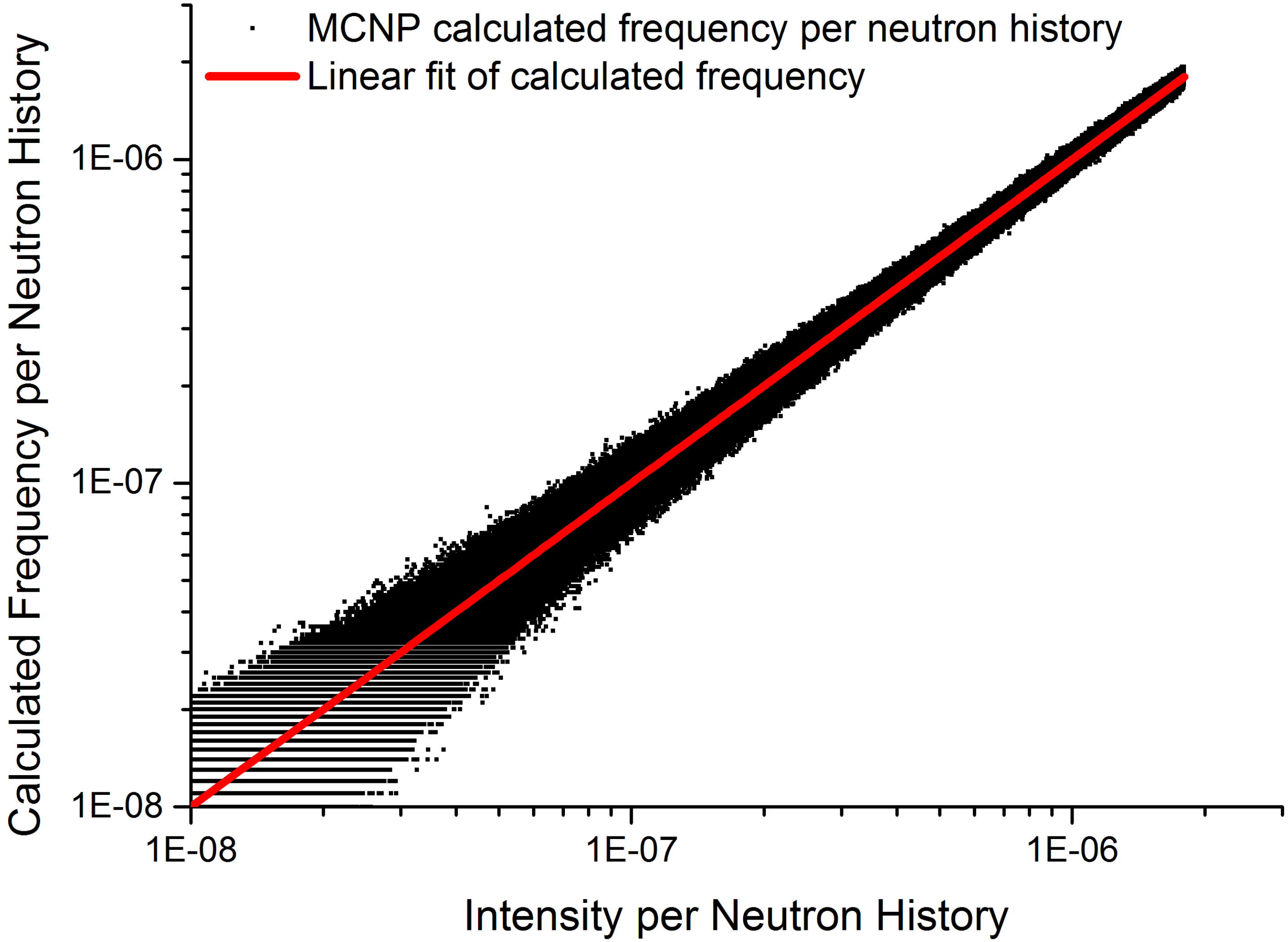
Check of the source points



Calculated frequency of the source positions compared with the original intensity

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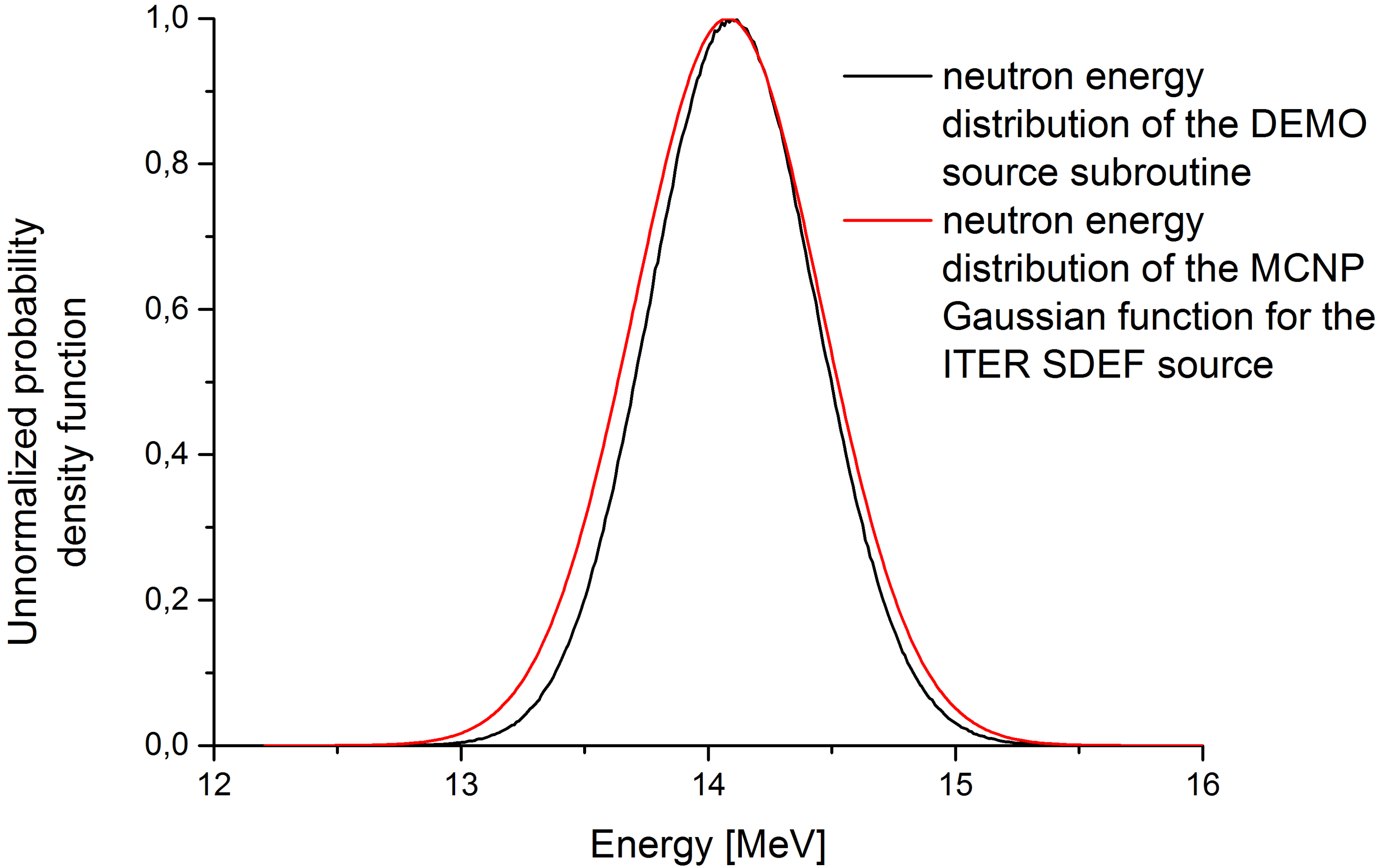
Check of the source points

- Normalized frequency compared with the original intensity has a linear behavior
- Statistical difference between frequency and intensity
- Frequency is an integer of $\frac{1}{Number\ of\ sampled\ Particles}$ → discrete steps visible in the low frequency region
- Linear fit at the curve has a gradient of 0.99997 and an error of 2.4×10^{-5}
  → very good agreement with expectations

Neutron energy distribution

- ITER SDEF source use Gaussian fusion spectrum from MCNP
- Energy distribution from DEMO source subroutine is available
- Comparison between Gaussian fusion spectrum of the ITER SDEF source and the spectrum of the DEMO source subroutine
- Test case: sample the DEMO energy calculation 10^7 times

Neutron energy distribution



Source development: Conclusion

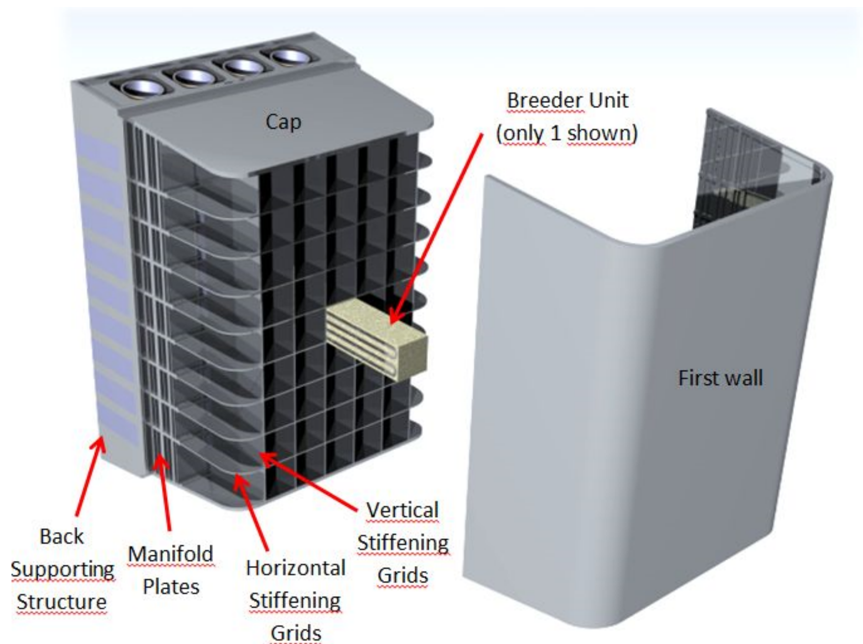
- Source development is finished
- Source tested to be reliable
- Cumulative probability of sampled and original data fits to each other
- Random sampling of XYZ-points in MCNP works and reproduce the original data
- Energy spectrum is calculated, tested and successfully compared to standard Gaussian spectrum → DEMO energy calculation will be used

CAD BASED GEOMETRY MODELING

Geometry modelling

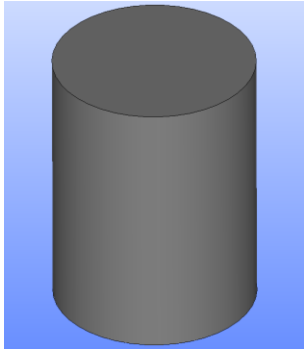
- Base: CAD data from IPP Greifswald
- Nuclear performance of the HELIAS reactor should be investigated
- Modelling of an arrangement of the components – like first wall, breeding blanket, shielding – in the blanket envelope

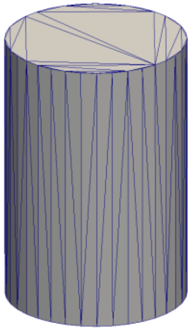
Helium Cooled Pebble Bed – DEMO Blanket Module



Geometry modelling

- Different approaches, all will be investigated:
  1. “Geometry translation approach” with KIT’s CAD to MCNP conversion tool McCad → fully developed way for Tokamak reactors
  2. Direct tracking of particles in CAD geometry by using DAG-MCNP (DAG = Direct Accelerated Geometry)
  3. Tracking of particles in unstructured meshes

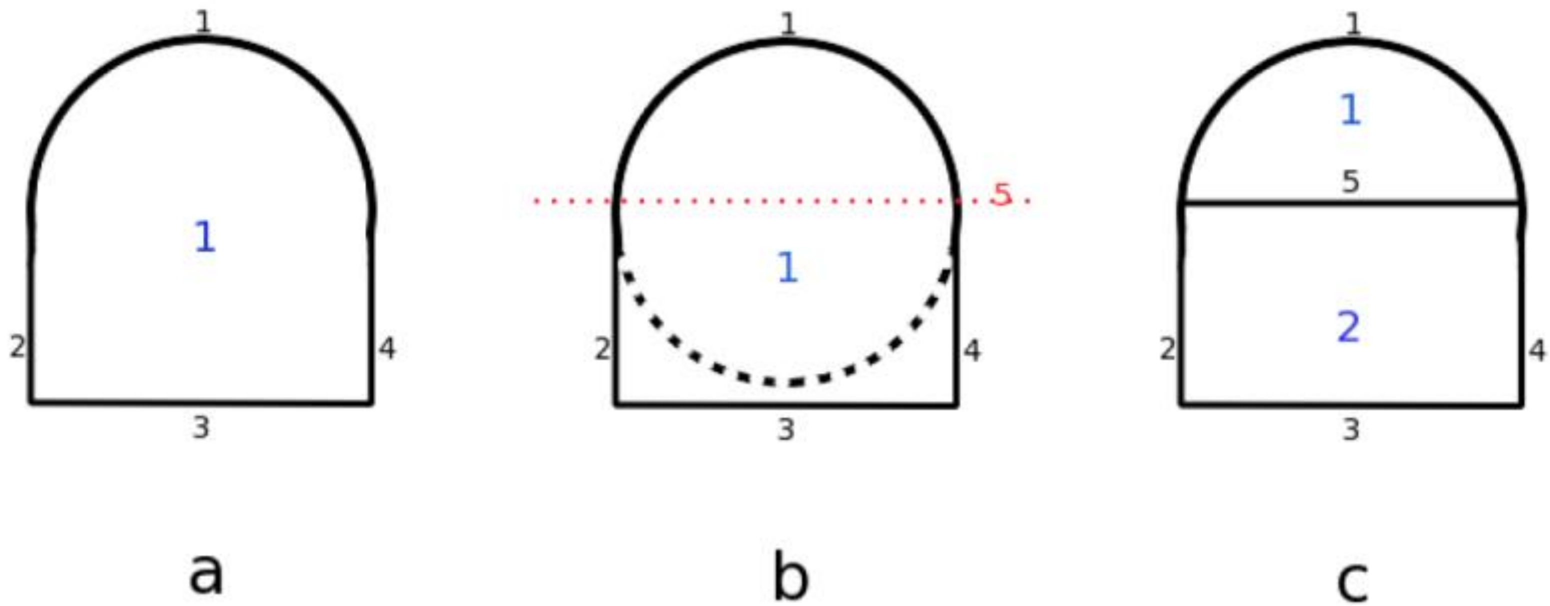




Geometry translation

- Model of the HELIAS reactor in Computer Aided Design (CAD)
- CAD Geometry:
  - boundary representation → representation of complicate surfaces like spline surfaces
- Constructive solid geometry (CSG):
  - Boolean combination of half spaces in MCNP
  - cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori
- Decomposition of the geometry is needed → time consuming task
- Conversion tool needed → McCad from KIT

Geometry translation



Source: [Gro09]

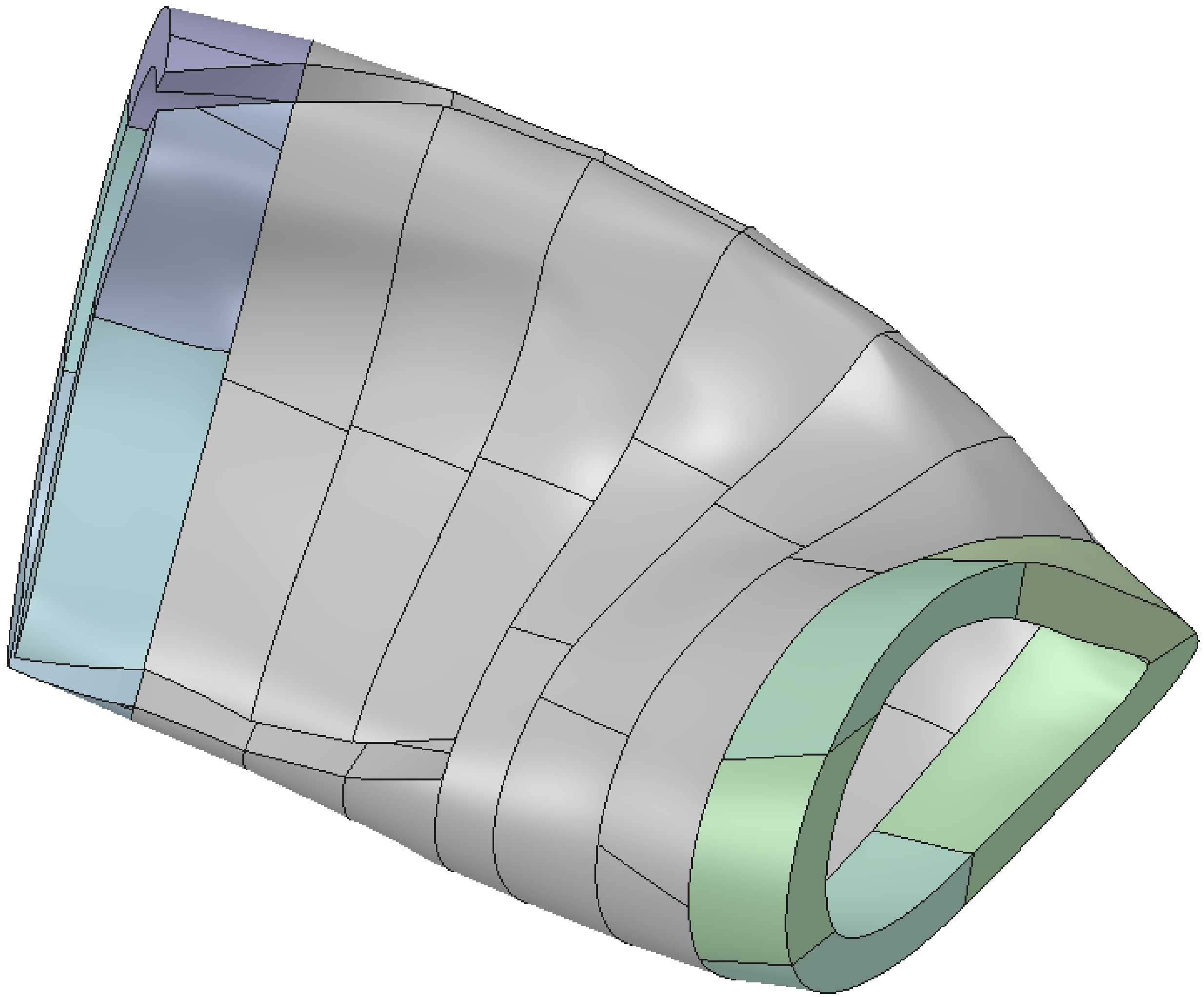
Surface from CAD description, not represented in MCNP

in MCNP defined as
(-1 5):(-5 2 3 -4)

or

and

HELIAS CAD Model



Blanket envelope model provided by IPP Greifswald

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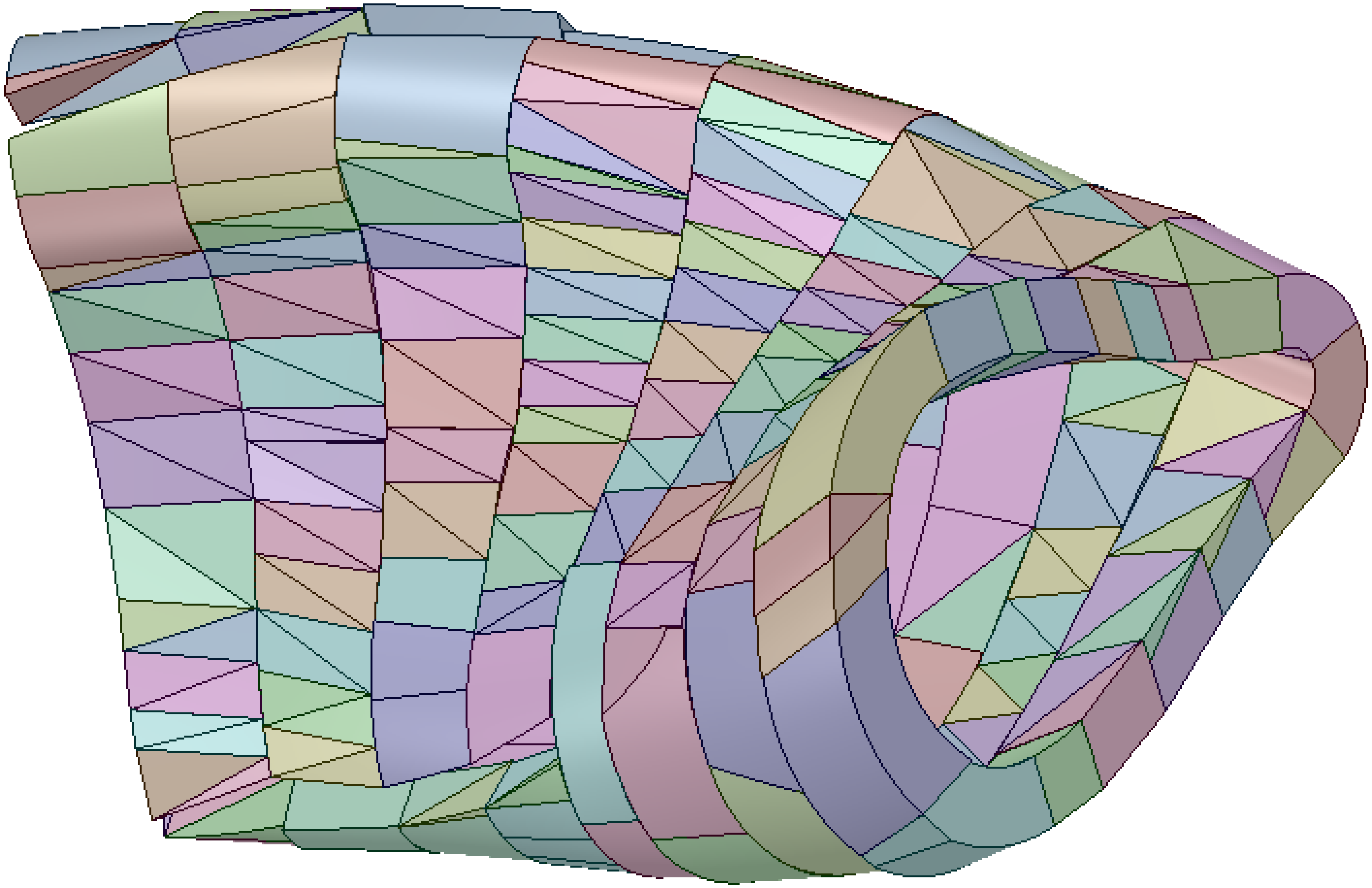
Decomposed HELIAS CAD Model



Simplified and decomposed blanket envelope model

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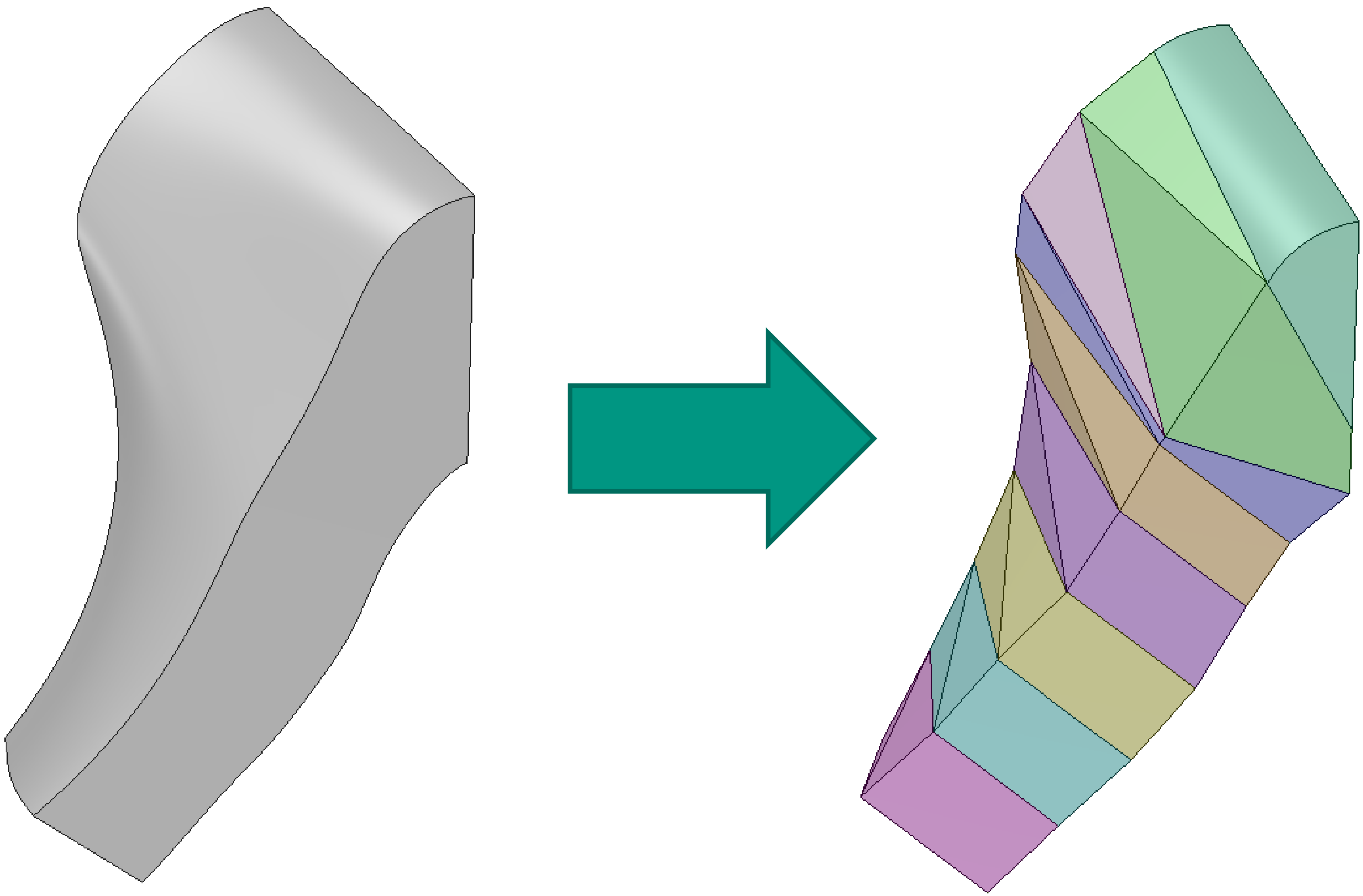
CAD Model decomposition



Decomposition of one blanket envelope module

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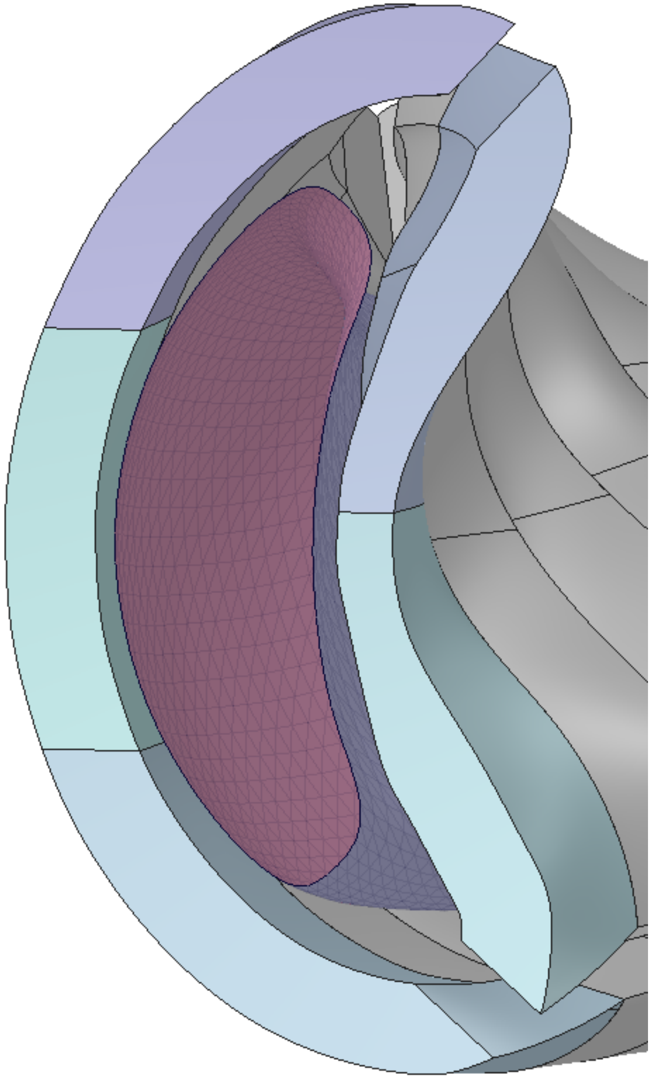
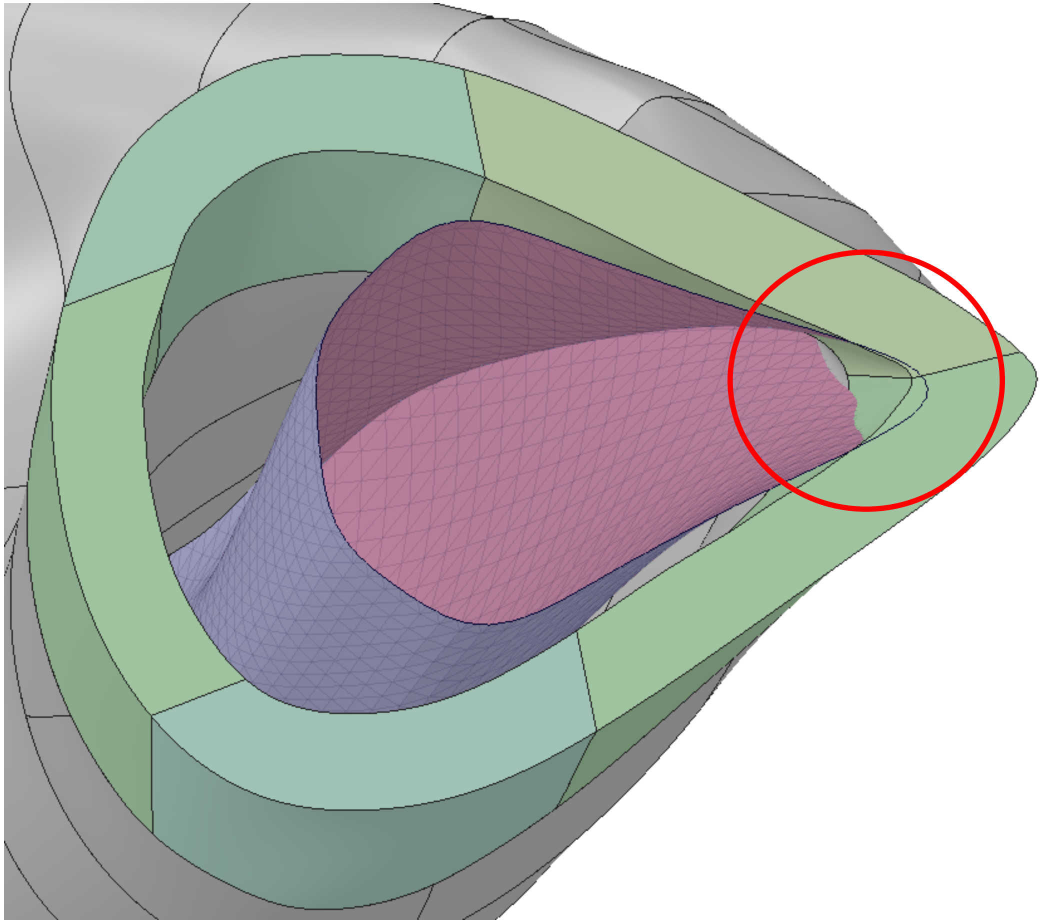
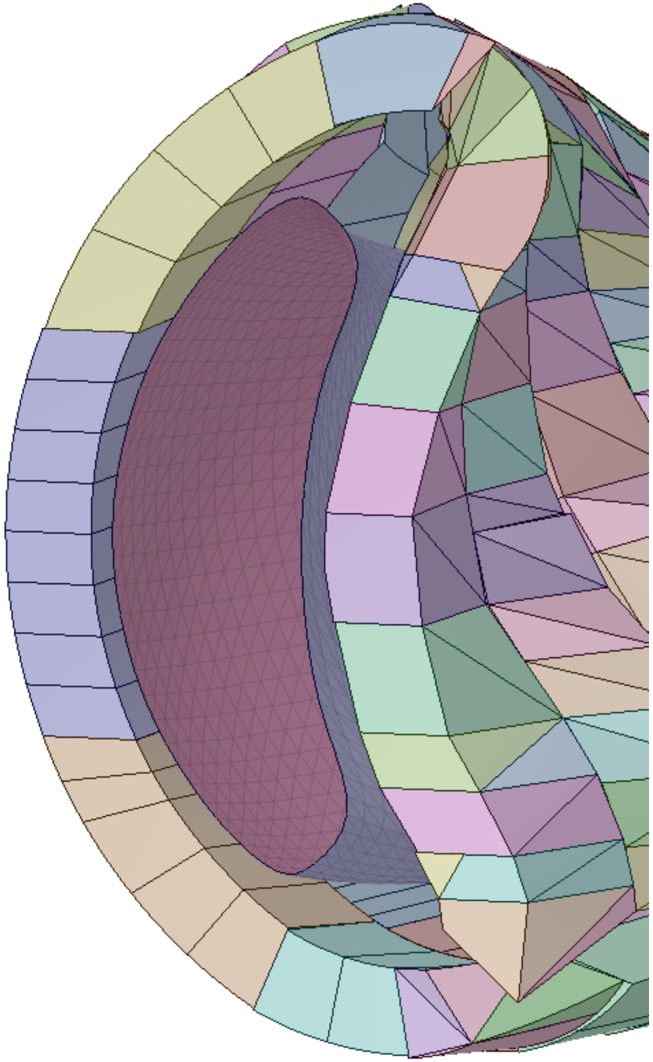
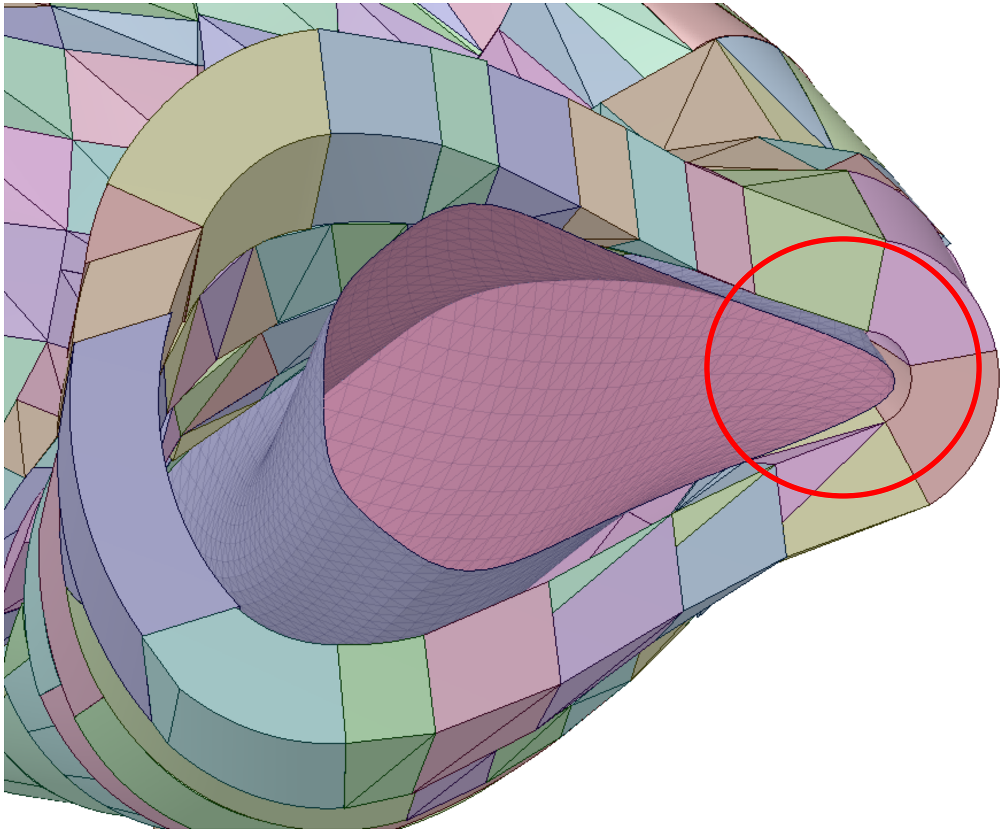
HELIAS model with last closed flux surface





Decomposed model with last closed flux surface





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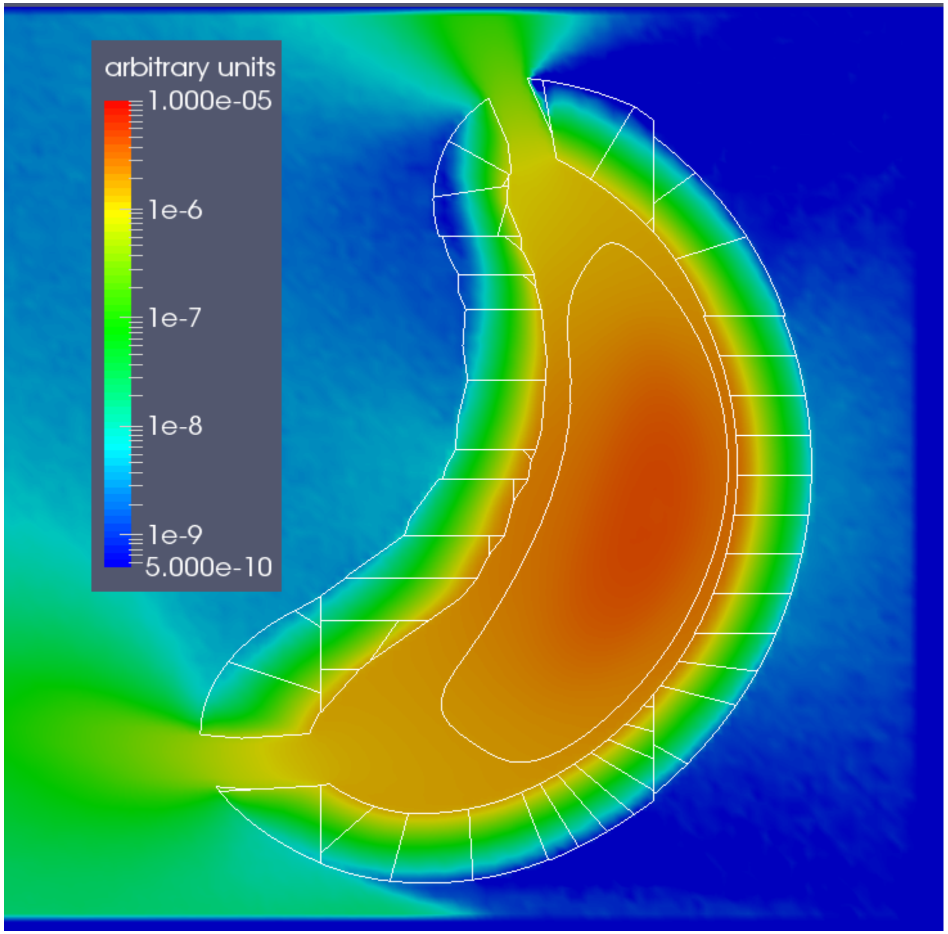
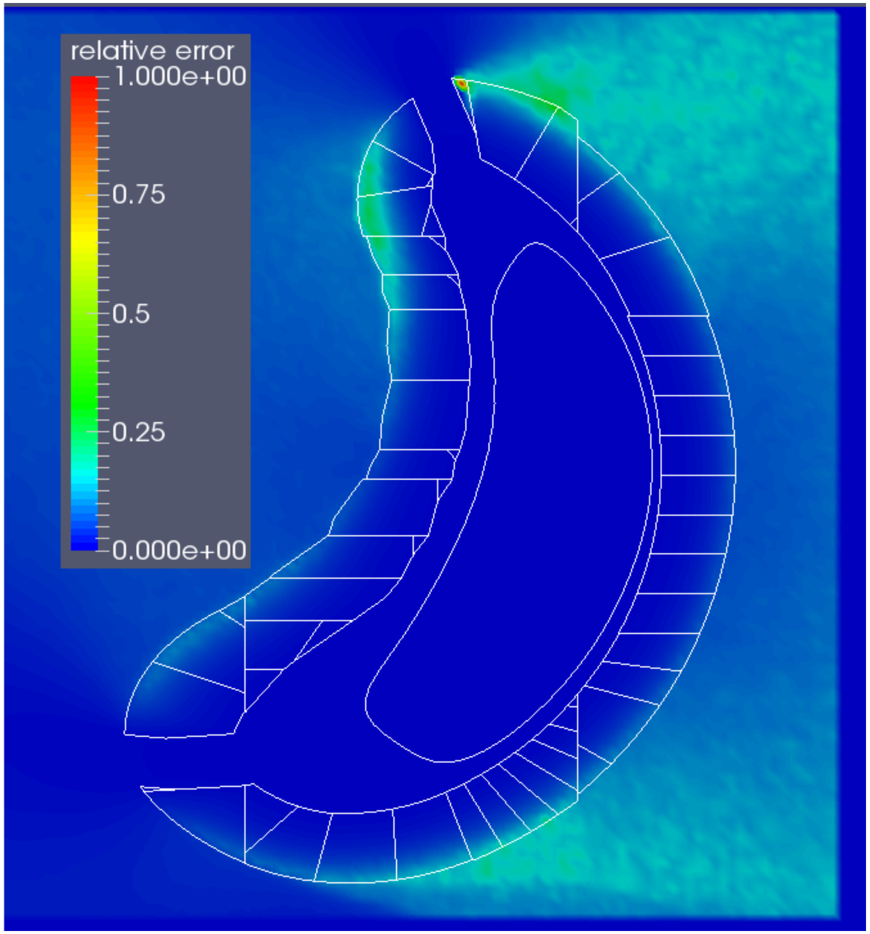
First calculations of the neutron flux

- Test if the converted model works together with the source subroutine
- Reflecting conditions at the reactor boundary
- Test material in the whole blanket
- Goal:
  - Check if the source and the blanket are in the same area
  - Check how many “lost particles” occur
  - Calculate the neutron flux distribution (“Meshtally”)

First calculations of the neutron flux





Distribution in plane perpendicular to the main axis. Geometry and last closed flux surface (white) is also shown.

Neutron flux distribution in arbitrary units.

Corresponding relative error.

First calculations of the neutron flux

- Preliminary result is obtained
- Run on single core for 4300 minutes (~72 h)
- Calculated particles: 1.05×10^9
- Lost particles: ~20% → Not accaptable
- Conclusion of the first test:
  - Blanket and source fit to each other
  - Too many “lost particles” → geometry repair and modification needed

CONCLUSION AND OUTLOOK

Conclusion

- Stellarator reactor has more complex geometry than Tokamak reactor
- 3-D Stellarator neutron source:

  → development and testing are successfully finished

- Geometry modelling:

  → preliminary model is finished and tested with source subroutine

Outlook

- Further development of the blanket model → first wall, breeding zone, manifolds, etc. are missing right now
- Nuclear design analyses:
  - Neutron wall loading
  - Nuclear heating
  - Neutron flux distribution
  - Shielding performance
  - Tritium breeding performance
  - Radiation damage

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Thank you all for listening.

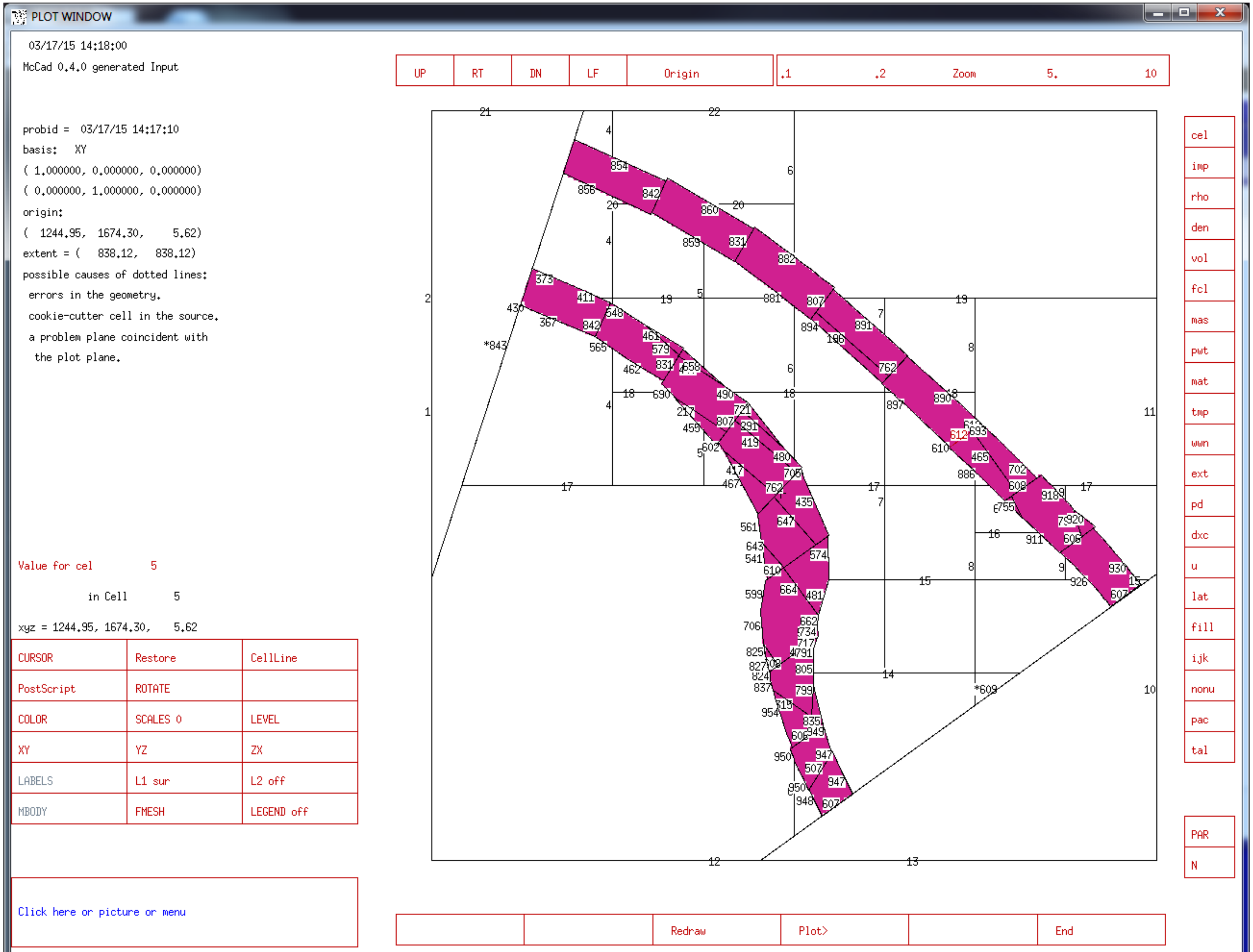
Any questions?

Source

- [Bei11] – C.D. Beidler, *Helical-Axis Advanced Stellarator (Helias) Reactors*, MFE Roadmapping in the ITER Era, September 2011
- [Gro09] – D. Große and H. Tsige-Tamirat, *Current Status of the CAD Interface Program for MC Particle Transport Codes McCad*, International Conference on Mathematics, Computational Methods & Reactor Physics (M&C 2009), Saratoga Springs, New York, May 3-7, 2009, on CD-ROM, American Nuclear Society, LaGrange Park, IL (2009)
- [IPP13] – Max-Planck-Institut für Plasmaphysik (IPP), *Fusion Basics Kernfusion Stand & Perspektiven*, www.ipp.mpg.de, March 2013
- [Lu13] – Lu L., et al., *Extended and Improved Capabilities of the CAD to MC Geometry Conversion Tool McCad,* ANS 2013 WINTER MEETING & TECHNOLOGY EXPO – Washington,D.C 10-14.Nov.2013
- [Mit09] – N. Mitchell, et al., *Status of the ITER magnets*, Fus. Eng. Des. 84 (2009)
- [Sch13] – F. Schauer, et al., *HELIAS 5-B magnet system structure and maintenance concept,* Fus. Eng. Des. 88 (2013)
- [Tea08] - X-5 MONTE CARLO TEAM, MCNP—A General Monte Carlo N-Particle Transport Code, Version 5, Volume I, MCNP Overview and Theory, LA-UR-03-1987, Los Alamos National Laboratory (Apr. 24, 2003; revised Feb. 1, 2008)

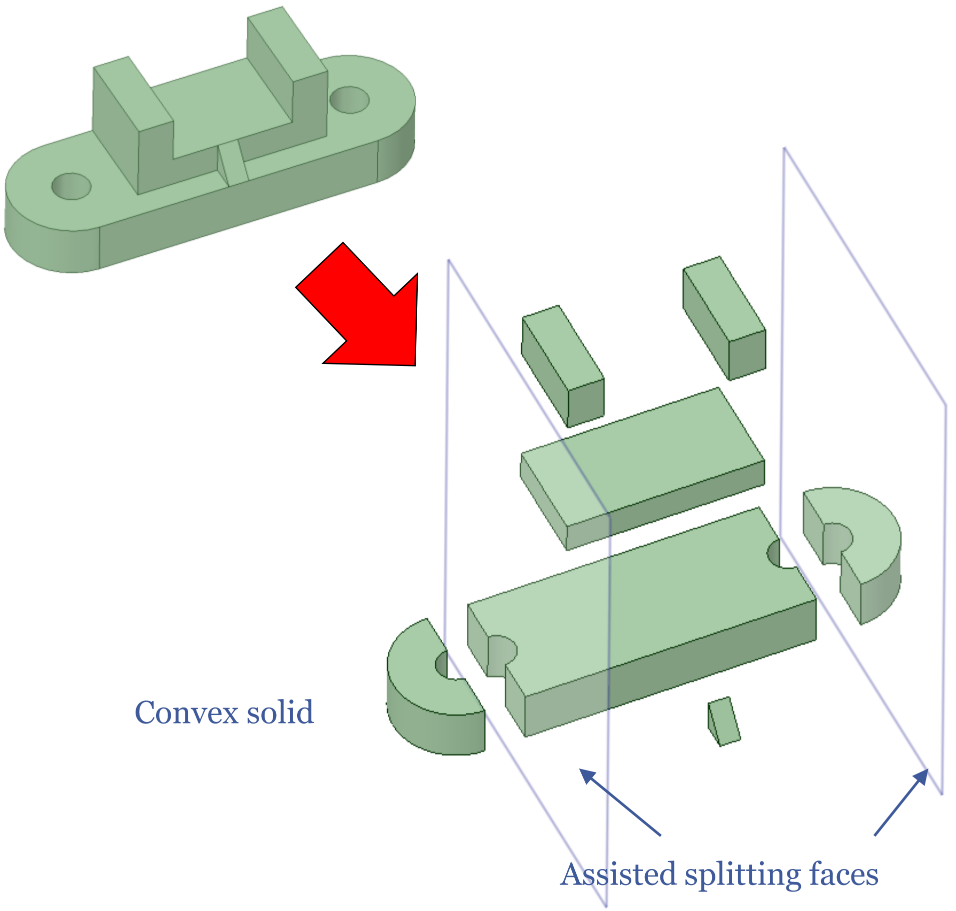
BACKUP

Geometry Cut XY-Plane

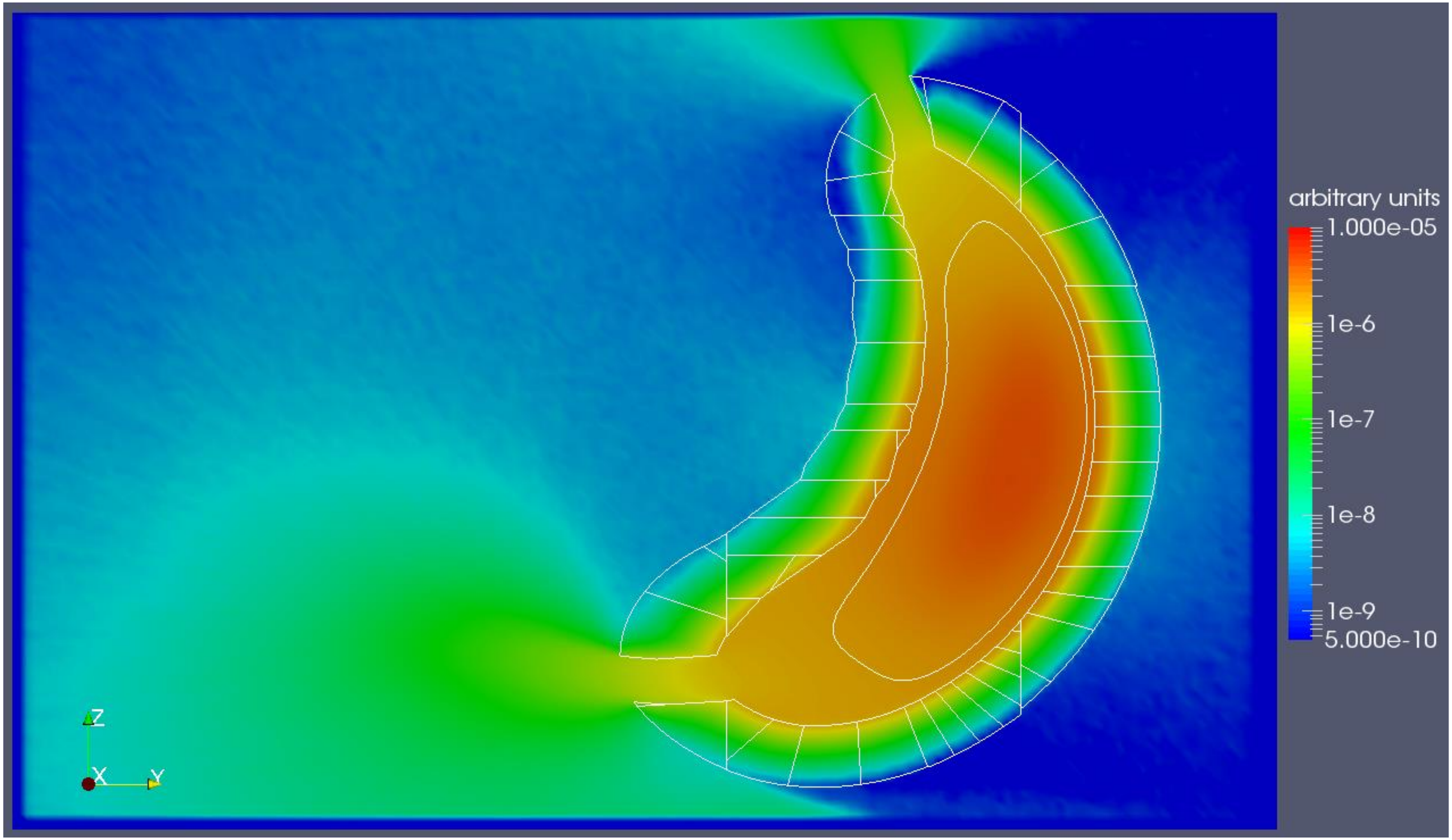


Geometry translation

Decomposition of a complex solid by McCad
Source: [Lu13]

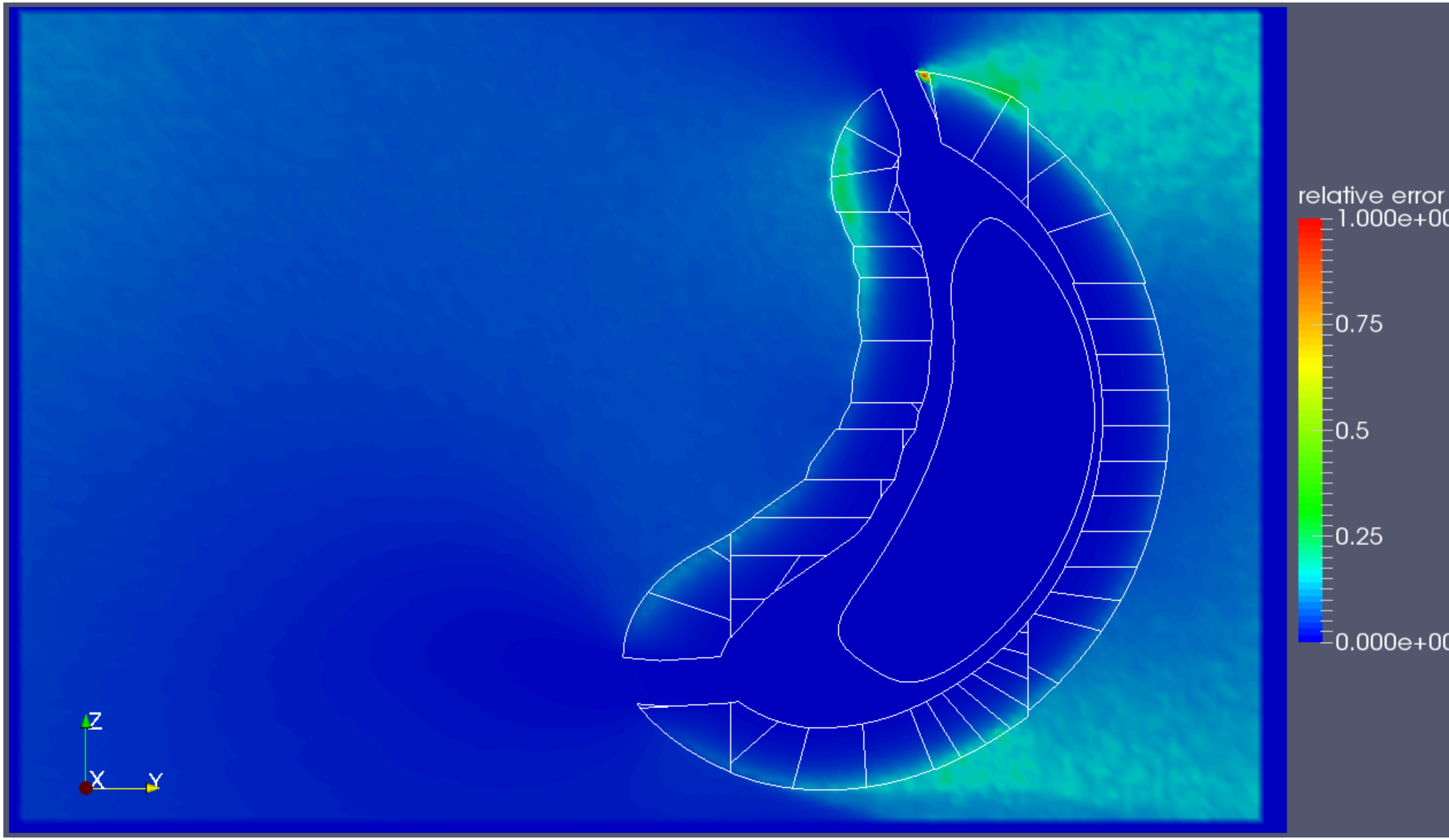


First calculations of the neutron flux



Result of the neutron flux distribution with arbitrary units in the perpendicular to the main axis; geometry and last closed flux surface are shown inside

First calculations of the neutron flux



Error of the neutron flux meshtally in the perpendicular to the main axis; geometry and last closed flux surface are shown inside

Comparison: Tokamak and Stellarator

Component	Tokamak	Stellarator
Coils	Identical planar shaped coils	Different non-planar modular coils
Plasma geometry	Elliptic-axisymmetric	Fully 3-D
Symmetry	Azimuthally symmetric	Discrete rotational symmetry (e.g. five field periods)
Operation length	Induced current through plasma → pulsed operation	No induced current through plasma → steady state operation